NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
VERMONT YANKEE NUCLEAR POWER STATION	0 5 0 0 0 2 7 1	1 OF 0 3

TITLE (4): REACTOR POWER/FLOW ANOMALY

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 9	1 6	8 4	8 4	0 2 1	0 0	1 0	1 6	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

POWER LEVEL (10):

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
James P. Pelletier, Plant Manager	AREA CODE 8 0 2 2 5 7 - 7 7 1 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
D	A C	S E P	G 0 8 0	N					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)   X NO

EXPECTED SUBMISSION DATE (15): MONTH / DAY / YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

While proceeding to full power operation on 9/11/84, it was observed that the relationship between core power and core flow rate was abnormal. A power level of only 95% was attained at 100% core flow compared to an expected level of 100%. An extensive review of plant operating parameters revealed that Reactor power did not increase (by approx. 50 megawatts) as expected at higher Reactor core flows (approx. 42 million pounds per hour). This was accompanied by an increase in vessel downcomer water temperature (approx. 2^{o} F). Investigations determined the most probable cause to be steam carryunder.

The reactor was shut down on 9/18/84. The reactor internals were examined and the shroud head bolts were observed to be latched, but not tightened. The reactor was subsequently reassembled during which proper bolt tension and shroud head seating were verified. Plant startup commenced on 10/1/84.

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NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

While proceeding to full power operation on 9/11/84, it was observed that the relationship between core power and core flow rate was abnormal. A power level of only 95% was attained at 100% core flow compared to an expected level of 100%. An extensive review of plant operating parameters revealed that Reactor power did not increase (by approx. 50 megawatts) as expected at higher Reactor core flows (approx. 42 Mlb/hr). This was accompanied by an increase in vessel downcomer water temperature (approx. 2° F).

On 9/16/84, a power reduction was initiated for turbine surveillance. Core flow was decreased at a rate of one Mlb/hr, power allowed to stabilize and data collected. Within the 42 to 44 Mlb/hr flow range, step changes were recorded upward for Reactor power as read on the APRM and downward for the recirculation temperature. The core power vs. core flow relationship was normal below 42 Mlb/hr. This, coupled with the data taken during the power increase, indicated the anomaly was flow related. As a result, an administrative limit was imposed limiting core flow to less than 41 Mlb/hr.

Evaluations carried out determined the potential sources to be: 1) increased carryunder due to incorrect water level in the vessel resulting in lower than normal water levels within the separator thereby resulting in increased leakage of hot fluid to the annulus region. 2) loosened shroud head bolts allowing the shroud head to lift when the delta pressure across it was adequate, 3) failure of the separator or dryer assemblies resulting in increased delta pressure, decreased water level within the separator, and increased leakage of hot fluid to the annulus region and 4) failure of the core spray sparger piping allowing hot fluid to enter the annulus region from within the shroud. The reactor was shutdown on 9/18/84 for internal inspections to determine the cause of the anomaly.

The inspection revealed no evidence of physical damage to the components inspected and no indications of wear or vibrations. All 36 shroud head bolts had approximately 1/4 to 5/8-inch clearance between the bolt head and the core shroud lugs. This clearance would allow the shroud to lift by approximately 3/16 to 7/16-inch (clearance minus the differential thermal expansion). General Electric calculations based on the noted anomalies confirmed that the amount of lift necessary to increase the carryunder fraction was consistent with the amount of lift allowed by the loosened shroud bolts. The reactor was subsequently reassembled during which proper bolt tension and shroud head seating were verified. Plant startup commenced on 10/1/84.

The root cause of the problem was concluded to be improper tightening of the bolts during reassembly from the refuel outage and/or incomplete seating of the shroud during installation. Procedural changes have been made to ensure proper shroud bolt tension and shroud head seating during installation of the separator assembly.

The following determinations have been made regarding potential consequences to the public. First, for this event, steam carryunder preheated the vessel downcomer water to approx. 2° F higher than normal thereby altering the anticipated neutron performance by approx. 0.2% reactivity. This is well within the less than 1% delta reactivity requirements of Technical Specification Section 3.3.6. Second, the nature of the as-found separator configuration invalidates the reactor system models used in the station safety analysis since these models do not assume a flow path from the upper core plenum volume directly to the vessel downcomer volume. However, the magnitude of the separator misconfiguration had a negligible impact on the results of the station safety analysis. Third, General Electric calculations have shown that the

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predicted seismic loads could have been satisfactorily reacted by the latched but not tightened shroud head bolts with a factor of safety of twice that required by the design basis. In addition, the shroud head could not move or tip enough to contact any other component. Thus the consequences of a seismic event would not have been increased and no damage would have occurred to the core shroud, the shroud head guide pins or the core spray line(s). Based on the above, there were no adverse consequences to the public health and safety. No previous similar occurrences have been reported.

VERMONT YANKEE NUCLEAR POWER CORPORATION

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GOVERNOR HUNT ROAD
VERNON, VERMONT 05354

VYV 84-511

October 16, 1984

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

REFERENCE: Operating License DPR-28
Docket No. 50-271
Reportable Occurrence No. LER 84-21

Dear Sirs:

As defined by 10CFR50.73, we are reporting the attached Reportable Occurrence as LER 84-21.

Very truly yours,

James P. Pelletier

James P. Pelletier
Plant Manager

RDP/drc

cc: Regional Administrator
USNRC Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

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